NRR-PMDAPEm Resource

From: Holden, Leslie E.:(GenCo-Nuc) [Leslie.Holden@exeloncorp.com]
Sent: Wednesday, January 29, 2014 8:22 PM
To: Wiebe, Joel
Cc: Borton, Kevin F:(GenCo-Nuc); Gullott, David M.:(GenCo-Nuc)
Subject: RE: Braidwood/Byron MUR Package for Proprietary and Factual Error Review - Part 1
Attachments: 20140129 MUR Package for Proprietary and Factual Error Review - Part 1 LEH.docx

Importance: High

Joel,

We appreciate you providing us the drafts and also the opportunity to comment.

Based on our review today we would like to offer you the following comments for your consideration:

Page 32/44 – Under the heading "License Response" the date should be June 23, 2011 (not 201**2**)
Page 33/44 – Item E 2nd ¶ - The bounding uncertainty provide in the Cameron Uncertainty Reports (ER-800 thru 803, LAR Attachments 8a thru d) was ±0.345%. Exelon added additional margin to the 0.345% uncertainty to limit the requested power increase to be 1.63%.
Page 34/44 – Item 4 "Reporting Deficiencies to the Manufacturer" - The wording is different than what we provided in the submittal. The wording we provided regarding Reporting Deficiencies to the Manufacturer (LAR, Attachment 5, Item 4 on page I-9 was "Deficiencies associated with the vendor's processes or equipment will be reported to the vendor as needed to support corrective action."
Page 35/44 – Under the heading "Items G and H," 2nd ¶ - In the last 2 sentence of the paragraph, the power values should be 3586.6 MWt (not 358**5**.6 MWt)
Page 40/44 – Under the heading of "In-Plant Operation" – There is a stipulation that if cross-checks between the UFM and other plant parameters related to feedwater identify "abnormal behavior" that it should be "identified to the NRC, the validity of the final commissioning test report should be confirmed, and the final commissioning test report should be updated as necessary to reflect the new information." We appreciate clarification of the NRC's intent for these actions as ongoing programmatic requirements.

These comments, as well as other minor editorial comments are indicated in "track changes" in Word "Review" mode on the attached.

Please feel free to give me a call if you have any questions.

Thank you,

Leslie

Leslie E. Holden
Senior Regulatory Engineer
Power Uprate

Cantera Office
4300 Winfield Road
Warrenville, **IL 60555**
Office: 630 657 3316 | Cell: 630 291 7329 | Fax: 630 657 4300
Leslie.Holden@ExelonCorp.com www.ExelonCorp.com

From: Wiebe, Joel [mailto:Joel.Wiebe@nrc.gov]
Sent: Wednesday, January 29, 2014 6:46 AM
To: Holden, Leslie E.:(GenCo-Nuc)
Subject: Braidwood/Byron MUR Package for Proprietary and Factual Error Review - Part 1

Leslie,

Here is Part 1 of the MUR package for Proprietary and Factual Error Review. The pagination issues are the result of my copy and paste actions. I didn't take the time to correct them.

Joel

Hearing Identifier: NRR_PMDA
Email Number: 1081

Mail Envelope Properties (242AFBA318413548869FC73CD15D0EDA065FF6)

Subject: RE: Braidwood/Byron MUR Package for Proprietary and Factual Error Review - Part 1
Sent Date: 1/29/2014 8:22:18 PM
Received Date: 1/29/2014 8:22:28 PM
From: Holden, Leslie E.:(GenCo-Nuc)

Created By: Leslie.Holden@exeloncorp.com

Recipients:
"Borton, Kevin F:(GenCo-Nuc)" <kevin.borton@exeloncorp.com>
Tracking Status: None
"Gullott, David M.:(GenCo-Nuc)" <David.Gullott@exeloncorp.com>
Tracking Status: None
"Wiebe, Joel" <Joel.Wiebe@nrc.gov>
Tracking Status: None

Post Office: EXCHM-CCC-22.exelonds.com

Files	**Size**	**Date & Time**
MESSAGE	2656	1/29/2014 8:22:28 PM
20140129 MUR Package for Proprietary and Factual Error Review - Part 1 LEH.docx	142012	

Options
Priority: High
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

Mr. Michael J. Pacilio
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. MF2418, MF2419, MF2420, AND MF2421)

Dear Mr. Pacilio:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 174 to Facility Operating License No. NPF-72 and Amendment No. 174 to Facility Operating License No. NPF-77 for the Braidwood Station, Units 1 and 2, respectively, and Amendment No. 181 to Facility Operating License No. NPF-37 and Amendment No. 181 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated June 23, 2011, as supplemented by letters dated August 25, November 1, and December 9, 2011; February 20, March 5, March 30 (two letters), April 27, May 16, June 26, August 8, September 13, and October 9, 2012; and July 5, September 5, October 8, October 24, November 13, and November 18, 2013.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457,
STN 50-454 and STN 50-455

Enclosures:
1. Amendment No. 174 to NPF-72
2. Amendment No. 174 to NPF-77
3. Amendment No. 181 to NPF-37
4. Amendment No. 181 to NPF-66
5. Safety Evaluation

cc w/encls: Distribution via Listserv

Mr. Michael J. Pacilio
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NOS. MF2418, MF2419, MF2420, AND MF2421)

Dear Mr. Pacilio:

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A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457,
STN 50-454 and STN 50-455

Enclosures:
1. Amendment No. 174 to NPF-72
2. Amendment No. 174 to NPF-77
3. Amendment No. 181 to NPF-37
4. Amendment No. 181 to NPF-66
5. Safety Evaluation

cc w/encls: Distribution via Listserv

See next page for DISTRIBUTION AND CONCURRENCE:

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RidsNrrPMBraidwood Resource
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RidsNrrDssStsb Resource

Accession No. ML13281A000 * via memo

OFFICE	LPL3-2/PM	LPL3-2/LA	NRR/AADB/BC*	NRR/AFPB/BC*	NRR/AHPB/BC*	NN/EEEB/BC*	NRR/EICB/BC*
NAME	JWiebe	SRohrer	TTate	AKlein	UShoop	JAnderson	GWilson
DATE	12/12/13	1/ /14	4/2/12	9/24/13	11/ 22 /11	5/3 /12	1/9/12
OFFICE	NRR/EMCB/BC*	NRR/ENPB/BC	NRR/EPTB/BC	NRR/ESGB/BC*	NRR/EVIB/BC*	NRR/SBPB/BC*	NRR/SCVB/BC
NAME	MMurphy	TLupold	AMcMurtray	GKulesa	HGonzalez	GCasto	RDenning
DATE	10/25/13	04/16/13	10/4/11	5/30/12	12/8/11	8/30/13	11/17/11
OFFICE	NRR/SNPB/BC*	NRR/SRXB/BC*	NRR/STSB/BC*	OGC	NRR/LPL2-1/BC	NRR/DORL/DD	LPL3-2/PM

OFFICE	LPL3-2/PM	LPL3-2/LA	NRR/AADB/BC*	NRR/AFPB/BC*	NRR/AHPB/BC*	NN/EEEB/BC*	NRR/EICB/BC*
NAME	AMendiola	CJackson	RElliott		TTate	MEvans	JWiebe
DATE	12/27/11	12/9/13	11/29/13	1/ /14	1/ /14	1/ /14	1/ /14

OFFICIAL RECORD COPY

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
License No. NPF-72

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated June 23, 2011, as supplemented by letters dated August 25, November 1, and December 9, 2011; February 20, March 5, March 30 (two letters), April 27, May 16, June 26, August 8, September 13, and October 9, 2012; and July 5, September 5, October 8, October 24, November 13, and November 18, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by changes to the Technical Specifications as indicated in the attachment to this license amendment and the following paragraphs the Facility Operating License No. NPF-72 is hereby amended to read as follows:

A. Paragraph 2.C.(1):

Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

B. Paragraph 2.C.(2)

Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 174 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michele G. Evans, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Facility Operating License

Date of Issuance:

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
License No. NPF-77

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated June 23, 2011, as supplemented by letters dated August 25, November 1, and December 9, 2011; February 20, March 5, March 30 (two letters), April 27, May 16, June 26, August 8, September 13, and October 9, 2012; and July 5, September 5, October 8, October 24, November 13, and November 18, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by changes to the Technical Specifications as indicated in the attachment to this license amendment and the following paragraphs the Facility Operating License No. NPF-77 is hereby amended to read as follows:

A. Paragraph 2.C.(1):

Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

B. Paragraph 2.C.(2)

Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 174 and the Environmental Protection Plan contained in Appendix B, both of which are attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michele G. Evans, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Facility Operating License

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NOS. 174 AND 174

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Facility Operating Licenses and Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-72	License NPF-72
Page 3	Page 3
License NPF-77	License NPF-77
Page 3	Page 3
TSs	TSs
Page 1.1-6	Page 1.1-6
Page 2.0-1	Page 2.0-1
Page 3.4.1-1	Page 3.4.1-1
Page 3.4.1-2	Page 3.4.1-2
Page 5.6-4	Page 5.6-4

(3) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 174 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provision of 10 CFR Section 50.54(s)(2) will apply.

material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Exelon Generation Company, LLC pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Exelon Generation Company, LLC pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 174 and the Environmental Protection Plan contained in Appendix B, both of which are attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provision of 10 CFR Section 50.54(s)(2) will apply.

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 181
License No. NPF-37

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated June 23, 2011, as supplemented by letters dated August 25, November 1, and December 9, 2011; February 20, March 5, March 30 (two letters), April 27, May 16, June 26, August 8, September 13, and October 9, 2012; and July 5, September 5, October 8, October 24, November 13, and November 18, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by changes to the Technical Specifications as indicated in the attachment to this license amendment and the following paragraphs the Facility Operating License No. NPF-37 is hereby amended to read as follows:

A. Paragraph 2.C.(1):

Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

B. Paragraph 2.C.(2)

Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 181 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michele G. Evans, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Facility Operating License

Date of Issuance:

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 181
License No. NPF-66

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated June 23, 2011, as supplemented by letters dated August 25, November 1, and December 9, 2011; February 20, March 5, March 30 (two letters), April 27, May 16, June 26, August 8, September 13, and October 9, 2012; and July 5, September 5, October 8, October 24, November 13, and November 18, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by changes to the Technical Specifications as indicated in the attachment to this license amendment and the following paragraphs the Facility Operating License No. NPF-66 is hereby amended to read as follows:

A. Paragraph 2.C.(1):

Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

B. Paragraph 2.C.(2)

Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 181 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michele G. Evans, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Facility Operating License

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NOS. 181 AND 181

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND STN 50-455

Replace the following pages of the Facility Operating License and Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-37	License NPF-37
Page 3	Page 3
License NPF-66	License NPF-66
Page 3	Page 3
TSs	TSs
Page 1.1-6	Page 1.1-6
Page 2.0-1	Page 2.0-1
Page 3.4.1-1	Page 3.4.1-1
Page 3.4.1-2	Page 3.4.1-2
Page 5.6-4	Page 5.6-4

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 181 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Deleted.

(4) Deleted.

(5) Deleted.

(6) The license shall implement and maintain in effect all provisions of the approved fire protection program as described in the licensee's Fire Protection Report, and as approved in the SER dated February 1987 through Supplement No. 8, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 181 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Deleted.

(4) Deleted.

(5) Deleted.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. NPF-72,

AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. NPF-77,

AMENDMENT NO. 181 TO FACILITY OPERATING LICENSE NO. NPF-37,

AND AMENDMENT NO. 181 TO FACILITY OPERATING LICENSE NO. NPF-66

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

BYRON STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-456, STN 50-457,

STN 50-454, AND STN 50-455,

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated June 23, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML111790026), as supplemented by letters dated August 25, November 1, and December 9, 2011; February 20, March 5, March 30 (two letters), April 27, May 16, June 26, August 8, September 13, and October 9, 2012; and July 5, September 5, October 8, October 24, November 13, and November 18, 2013 (ADAMS Accession Nos. ML11255A332, ML113050427, ML113430811, ML12052A113, ML12066A052, ML12093A242, ML12096A051, ML12121A496, ML12138A091, ML12178A294, ML12222A037, ML12258A330, ML12285A087, ML13186A178, ML13248A519, ML13281A819, ML13298A477, ML13318A232, and ML13324A94, respectively), Exelon Generation Company, LLC (Exelon, the licensee) requested changes to the technical specifications (TSs) and facility operating licenses for Braidwood Station (Braidwood), Units 1 and 2, and Byron Station (Byron), Unit Nos. 1 and 2. Portions of the letters dated June 23, 2011 and December 9, 2012 contain sensitive unclassified non-safeguards information and, accordingly, have been withheld from public disclosure. The supplemental letters dated August 25, November 1, and December 9, 2011; February 20, March 5, March 30 (two letters), April 27, May 16, June 26, August 8, September 13, and October 9, 2012; and July 5, September 5, October 8, October 24, November 13, and November 18, 2013, provided additional clarifying information that did not expand the scope of the initial application and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the Federal Register on December 6, 2011 (77 FR 28630).

Enclosure

The proposed changes would increase the maximum steady-state reactor core power level from 3586.6 megawatts thermal (MWt) to 3645 MWt, which is an increase of approximately 1.63 percent. In addition to the above changes, a revised steam generator tube rupture (SGTR) and margin to overfill (MTO) analysis was submitted for NRC approval and is evaluated in this safety evaluation. This revised analysis was performed as the MTO values in the current analysis of record (AOR) are unacceptably small and revisions to the analysis assumptions are necessary. Specifically, the licensee proposes to revise the TSs:

- Section 2.1.1 to modify the departure from nucleate boiling (DNB) ratio and use of DNB correlations;
- Limiting Condition for Operation (LCO) 3.4.1 to modify the reactor coolant system (RCS) total flow rate for measurement uncertainty recapture uprated power conditions; and
- TS 5.6.5 to add analytical methods used to determine the core operating limit.

2.0 REGULATORY EVALUATION

2.1 Measurement Uncertainty Recapture (MUR) Power Uprate (PU)

Nuclear power plants are licensed to operate at a specified maximum core thermal power, often called RTP [reactor thermal power]. In Section 50.46 of Title 10 to the *Code of Federal Regulations* (10 CFR), "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," the NRC requires that emergency core cooling system (ECCS) performance under loss of coolant accident (LOCA) conditions be evaluated and that the estimated performance satisfy certain criteria. Licensees may conduct an analysis that "realistically describes the behavior of the reactor system during a LOCA" (often termed a "best-estimate analysis"), or they may develop a model that conforms with the requirements of Appendix K, "ECCS Evaluation Models," 10 CFR Part 50. Most ECCS evaluations are based on Appendix K requirements.

Appendix K, "[Emergency Core Cooling System] ECCS Evaluation Models," of 10 CFR Part 50, formerly required licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant accident (LOCA) and ECCS analyses. This requirement was included to ensure that instrumentation uncertainties were adequately accounted for in the safety analyses. In practice, many of the design bases analyses assumed a 2 percent power uncertainty, consistent with 10 CFR Part 50, Appendix K. The NRC staff considers the assumed power level to be an input parameter of the ECCS evaluation.

A change to the Commission's regulations at 10 CFR Part 50, Appendix K, was published in the Federal Register on June 1, 2000 (65 FR 34913), which became effective July 31, 2000. This change allows licensees to use a power level less than 1.02 times the RTP for the LOCA and ECCS analyses, but not a power level less than the licensed power level, based on the use of state-of-the art feedwater flow measurement devices that provide a more accurate calculation of power. Licensees can use a lower uncertainty in the LOCA and ECCS analyses provided that the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. As there continues to be substantial conservatism in other Appendix K requirements, sufficient margin to ECCS performance in the event of a LOCA is preserved.

However, this change to 10 CFR 50, Appendix K, did not authorize increases in licensed power levels for individual nuclear power plants. As the licensed power level for a plant is contained in

its operating license, licensees seeking to raise the licensed power level must submit a license amendment request (LAR) which must be reviewed and approved by the NRC staff. Braidwood, Units 1 and 2, and Byron, Unit Nos. 1 and 2, are currently licensed to operate at a maximum power level of 3586.6 MWt, which includes a 2 percent margin in the ECCS evaluation model to allow for uncertainties in rated thermal power (RTP) measurement. The LAR proposes to reduce this uncertainty to 0.37 percent.

2.2 Applicable Guidance

Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," (Reference 12), describes a method acceptable to the NRC staff for complying with the NRC regulations for assuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits. The method described in RG 1.105 for combining instrument uncertainties can be used for combining the uncertainties associated with the secondary calorimetric calculation. This allows licensees to justify a power uprate with reduced margin between the licensed power level and the power level assumed in the ECCS analysis by using more accurate instrumentation to calculate the reactor thermal power.

RG 1.105, Revision 3, endorses the Instrument Society of America (ISA) 67.04, Section 4.4.1, "Square-root-sum-of-squares method (SRSS)" which states:

> It is acceptable to combine uncertainties that are random, normally distributed, and independent by the SRSS method. When two independent uncertainties, (± a) and (± b), are combined by this method, the resulting uncertainty is (± c), where $c = (a^2 + b^2)^{1/2}$.

This standard in Section 4.4.2, "Algebraic method" states:

> It is acceptable to combine uncertainties that are not random, not normally distributed, or are dependent by the algebraic method. In this method, the combination of two dependent uncertainties, (+a, -0) and (+0, -b), results in a third uncertainty distribution with limits (+a, -b).

In the review of uncertainties in determining a trip setpoint and its allowable values, the NRC staff typically uses 95/95 tolerance limits as an acceptable criterion, i.e., a 95 percent probability that the constructed limits contain 95 percent of the population of interest for the surveillance interval selected.

The NRC determined that the two topical reports Cameron Engineering Report (ER)-80-P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM [leading edge flow measurement] Check™ System," March 1997, and Cameron ER-157-P, "Supplement to Engineering Report ER-80-P: Bases for a PU with the LEFM CheckPlus™ or LEFM CheckPlus™ (ADAMS Accession Nos. ML003670328 and ML102950252, respectively), that describe the Cameron LEFM CheckPlus System for the measurement of feedwater flow -- are an acceptable way of conforming to the regulations. The NRC's determination is documented in the NRC staff's safety evaluations dated March 8, 1999 (ADAMS Accession No. ML11353A016) and August 16, 2010 (ADAMS Accession No. ML102950252), respectively.

Revision 8 of ER-157- P, states that:

> [t]he redundancy inherent in the two measurement planes of an LEFM CheckPlus also makes this system more resistant to component failures,

when compared to the Check.

> For any single component failure, continued operation at a power greater than that prior to the uprate can be justified with a CheckPlus system … since the system with the failure is no less than an LEFM Check.

As stated in the NRC staff's SE dated August 16, 2010 (ADAMS Accession No. ML1029502520, Licensees referencing ER-157-P, Revision 8, must ensure compliance with the following limitations and conditions:

1. Continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time are plant-specific and must be acceptably justified.

2. The only mechanical difference that potentially affects the quoted statement is that the CheckPlus has 16 transducer housing interfaces with the flowing water, whereas, the LEFM Check has 8. Consequently, a CheckPlus operating with a single-failure that is assumed to disable one plain of transducers is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate as stated. An acceptable quantification method is to establish the effect in an acceptable test configuration such as can be accomplished at the Alden Research Laboratories (ARL).

As stated in the NRC staff's SE dated August 16, 2010 (ADAMS Accession No. ML1029502520, Licensees can reference ER-157-P, Revision 8, in their applications for MUR PUs for LEFM Check or LEFM CheckPlus system subject to the following additional qualifications:

1. A CheckPlus operating with a single-failure is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate using the degraded CheckPlus at an increased uncertainty.

2. An applicant with a comparable geometry can reference the Section 3.2.1 in ER-157-P, Revision 8, finding to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test results do not apply to a Check and downstream effects with use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable Alden Laboratory tests.

3. An applicant that requests a MUR with the upstream flow straightener configuration discussed in Section 3.2.2 should provide justification for claimed CheckPlus uncertainty that extends the justification provided in a letter from Caldon dated March 19, 2010, "Documentation to support review of Caldon ER-157-P, Revision 8," (ADAMS Accession No. ML100840025). Since this evaluation does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener.

4. An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of the uncertainties or, alternatively, should ensure that their calculations provide margin sufficient to cover the differences shown in Figure 1 of a letter dated March 18, 2010 (ADAMS Accession No. ML100820167).

2.3 Method of NRC Staff Review

The NRC staff reviewed the licensee's application to ensure that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. The purpose of the NRC staff's review is to ensure that instrumentation uncertainties were adequately accounted for in the safety analyses and to determine whether the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. - The NRC issued Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications" (ADAMS Accession No. ML013530183), to provide guidance to licensees on the scope and detail of the information that should be provided to the NRC staff for MUR PU requests. While RIS 2002-03 does not set forth NRC requirements, its use aids licensees in the preparation of their MUR PU submittal, while also providing guidance to the NRC staff for the conduct of its review.

Comment [LEH1]: Not sure if this is what was meant to be said.

The NRC staff's review of the LAR focused on the following information suggested in Attachment 1 to the RIS 2002-03 to determine whether the proposed MUR PU meets applicable NRC requirements:

- A detailed description of the plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique (Section I – Attachment 1, RIS 2002-03);
- A detailed discussion for each accident and/or transient for which the existing analyses of record bound plant operation at the proposed uprated power level and those analyses that are not covered by the reload methodology for the plant (Sections II and III);
- A discussion of the effect of the power uprate on the structural integrity of major plant components (Section IV);
- A discussion of the effect of the power uprate on electrical equipment (Section V);
- A discussion of the effect of the power uprate on major plant systems (Section VI);
- Other (Section VII), and;
- Changes to technical specifications, protection system settings, and emergency system settings (Section VIII).

The licensee stated in its application that its LAR was submitted consistent with the guidance of RIS 2002-03.

2.4 Proposed MUR PU at Braidwood, Units 1 and 2 and Byron, Unit Nos. 1 and 2

The proposed amendment is based on the use of the Cameron LEFM CheckPlus ultrasonic, multi-path, transit time flowmeter system that would decrease the uncertainty in the measurement of feedwater flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.37 percent.

The proposed PU is based on a redistribution of analytical margin originally required of ECCS evaluation models performed in accordance with the requirements set forth in 10 CFR Part 50, Appendix K, and "ECCS Evaluation Models." Appendix K mandated consideration of 102 percent of the licensed power level for ECCS evaluation models of light-water reactors. The NRC approved a change to the requirements of 10 CFR 50, Appendix K, on June 1, 2000. The change provided licensees with the option of maintaining the 2 percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation based on the accounting of uncertainties due to instrumentation error.

In existing nuclear power plants, the neutron flux instrumentation continuously indicates the reactor core thermal power. This instrumentation must be periodically calibrated to accommodate the effects of fuel burnup, flux pattern changes, and instrumentation setpoint drift. The reactor core thermal power generated by a nuclear power plant is determined by steam plant calorimetry, which is the process of performing a heat balance around the nuclear steam supply system (called a calorimetric). The accuracy of this calculation depends primarily upon the accuracy of feedwater flow rate and net enthalpy measurements. As such, an accurate measurement of feedwater flow rate and temperature is necessary for an accurate calibration of the nuclear instrumentation. Of the two parameters, flow rate and temperature, the most important in terms of calibration sensitivity is the FW flow rate.

The originally installed instruments for measuring feedwater flow rate in existing nuclear power plants were usually a venturi or a flow nozzle, each of which generates a differential pressure proportional to the feedwater velocity in the pipe. Of the two, the venturi was the most widely used because of relatively low head loss. However, error in determination of flow rate is introduced due to venturi fouling and, to a lesser extent, flow nozzle fouling, the transmitter, and the analog-to-digital converter.[1]

Because of the desire to reduce flow instrumentation uncertainty to enable operation of the plant at a higher power while remaining within the licensed rating, the industry assessed alternate flow rate measurement techniques and found that ultrasonic flow meter (UFMs) are a viable alternative. UFMs are based on computer-controlled electronic transducers that do not have differential pressure elements that are susceptible to fouling. Caldon, Inc. (now part of Cameron Measurement Systems) developed a UFM called a "leading edge flow meter" and named it the LEFM Check system. It followed this with the LEFM CheckPlus system, which consists essentially of two Check systems in the same spool piece and provides a more accurate FW flow measurement than the Check system. Both of these UFMs have demonstrated better measurement accuracies than the differential pressure type instruments and provide on-line verification to ensure that the UFM is operating within its uncertainty bounds.

Caldon submitted an engineering topical report (TR) ER-80-P, in March 1997, that describes the LEFM, includes calculations of power measurement uncertainty using a Check system in a typical two-loop pressurized-water reactor or a two-feedwater-line boiling-water reactor (BWR) and provides guidance for determining plant-specific power calorimetric uncertainties. The NRC staff approved this report on March 8, 1999 (ADAMS Accession No. ML11353A016), for a Comanche Peak Steam Electric Station exemption to the 2 percent uncertainty requirement in Appendix K to 10 CFR Part 50 and approved a 1 percent uprate for using the LEFM at Comanche Peak Steam Electric Station. Following publication of the amendment to Appendix K

[1] "Venturi" will generally be used in the remainder of this document to reference both venturis and flow nozzles.

(Section I.A) (Federal Register (65 FR 34921) dated June 1, 2000) that allowed for an uncertainty less than 2 percent, Caldon submitted a supplement in May 2000 to ER-80-P and ER-160-P, that the NRC staff approved in Reference D for up to 1.4 percent uprate. Subsequently, the NRC staff approved ER-157-P, Revision 5, for up to a 1.7 percent uprate using the CheckPlus system, and later approved ER-157-P, Revision 8. Revision 8 corrects minor errors in Revision 5, provides clarifying text, and incorporates revised analyses of coherent noise, non-fluid delays, and transducer replacement. It also adds two new appendices, Appendix C and Appendix D, which describe the assumptions and data that support the coherent noise and transducer replacement calculations, respectively.

Byron and Braidwood were originally designed with feedwater flow and temperature instrumentation consisting of venturis, differential pressure transmitters, and thermocouples. Modifications required for the MUR PU include installation of the CheckPlus system. Existing feedwater flow and temperature instrumentation will be retained and used for comparison monitoring of the LEFM system and as a backup feedwater flow measurement when needed.

The feedwater flow measurement system to be permanently installed is a Cameron LEFM CheckPlus ultrasonic eight-path transit time flowmeter. As discussed above, the CheckPlus design is addressed in ER-80-P, ER-160-P, and ER-157-P, ~~that~~ which have been approved by the NRC. It will be used for continuous calorimetric power calculation determination by providing feedwater mass flow and temperature input data to the plant computer system that is used for automated performance of the calorimetric power calculations.

The CheckPlus system consists of one flow element (spool piece) installed in each of the steam generator (SG) feedwater flow headers. The feedwater piping configurations are stated to have been explicitly modeled as part of the CheckPlus meter factor and accuracy assessment testing performed at ARL. The planned installation location of each CheckPlus is stated to conform to the applicable requirements in Cameron's Installation and Commissioning Manual and Cameron ER-80-P and ER-157-P, and the bounding uncertainty analysis is stated to be addressed by ER-720.

The LEFM instrumentation has been installed in each of the four units as follows:

- Braidwood Unit 1 Spring 2012 (during refueling outage (RFO) A1R16);
- Braidwood Unit 2 Spring 2011 (during RFO A2R15);
- Byron Unit 1 Spring 2011 (during RFO B1R17); and,
- Byron Unit 2 Fall 2011 (during RFO B2R16)

3.0 TECHNICAL EVALUATION

3.1 Safety Systems

3.1.1 Feedwater Flow Measurement Technique and Power Measurement Uncertainty (RIS 2002-03, Attachment 1, Section I)

3.1.1.1 Instrumentation and Control

Regulatory Evaluation

Section 50.36(c)(1)(ii)(A) of 10 CFR Part 50, “Technical specifications” requires, in part, that, where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

General Design Criterion (GDC) 13 of Appendix A, 10 CFR 50, "Instrumentation and Control," requires among other things, that instrumentation be provided to monitor variables and systems and that controls be provided to maintain these variables and systems within prescribed operating ranges.

General Design Criterion 20, of Appendix A, "Protection System Functions," requires, among other things, that the protection system be designed to initiate operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded.

In RG 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation,” a method acceptable to the NRC staff is described for complying with the NRC regulations for assuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits. The method described in RG 1.105 for combining instrument uncertainties can be used for combining the uncertainties associated with the secondary calorimetric calculation. This allows licensees to justify a power uprate with reduced margin between the licensed power level and the power level assumed in the ECCS analysis by using more accurate instrumentation to calculate the reactor thermal power.

As discussed in Section 2.2, the NRC approved two topical reports ER-80-NP-A and Cameron ER-157-NP-A as an acceptable way of conforming to the regulations. When it originally approved these reports, the NRC established four criteria for licensees to address. The criteria are as follows:

Comment [LEF2]: The other references refer to the “Proprietary” versions [P] instead of the “Non-Proprietary” versions [NP}.

Comment [LEF3]: The other references refer to the “Proprietary” versions [P] instead of the “Non-Proprietary” versions [NP}.

- Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.

- For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed instrumentation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analyses and assumptions set forth in ER-80-P.

- Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

- Licensees for plant installations where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), should provide additional justification for use. This justification should show that the meter installation is either independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher

Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Technical Evaluation

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system. This calculation is called “secondary calorimetric” for a pressurized-water reactor (PWR). The accuracy of this calculation depends primarily on the accuracy of feedwater flow and feedwater net enthalpy measurements. Feedwater flow is the most significant contributor to the core thermal power uncertainty. A more accurate measurement of this parameter will result in a more accurate determination of core thermal power.

Feedwater flow rate is typically measured using a venturi. This device generates a differential pressure proportional to the feedwater velocity in the pipe. Because of the high cost of calibrating the venturi and the need to improve flow instrumentation measurement uncertainty, the industry evaluated other flow measurement techniques and found the Cameron LEFM CheckPlus UFMs to be viable alternatives.

LEFM Technology and Measurement

The Cameron LEFM CheckPlus system uses a transit time methodology to measure fluid velocity. The basis of the transit time methodology for measuring fluid velocity and temperature is that ultrasonic pulses transmitted through a fluid stream travel faster in the direction of the fluid flow than opposite the flow. The difference in the upstream and downstream traversing times of the ultrasonic pulse is proportional to the fluid velocity in the pipe, and the temperature is determined using a pre-established correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure.

The system uses multiple diagonal acoustic paths instead of a single diagonal path, allowing velocities measured along each path to be numerically integrated over the pipe cross-section to determine the average fluid velocity in the pipe. This fluid velocity is multiplied by a velocity profile correction factor, the pipe cross-section area, and the fluid density to determine the feedwater mass flow rate in the piping. The mean fluid density may be obtained using the measured pressure and the derived mean fluid temperature as an input to a table of thermodynamic properties of water. The velocity profile correction factor is derived from calibration testing of the LEFM CheckPlus system in a plant-specific piping model at a calibration laboratory.

The Cameron LEFM CheckPlus system uses 16 transducers, eight each in two orthogonal planes of the spool piece. In the Cameron LEFM CheckPlus system, when the fluid velocity measured by an acoustic path in one plane is averaged with the fluid velocity measured by its companion path in the second plane, the transverse components of the two velocities are canceled and the result reflects only the axial velocity of the fluid. This makes the numerical integration of four pairs of averaged axial velocities and computation of volumetric flow inherently more accurate than a result obtained using four acoustic paths in a single plane. Also, because there are twice as many acoustic paths in the CheckPlus System, than in the Check System, and there are two independent clocks to measure the transit times, errors associated with uncertainties in path length and transit time measurements are reduced.

The NRC staff's review in the area of instrumentation and control covers the proposed plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique. The staff conducted its review to confirm that the licensee's implementation of the proposed feedwater flow measurement device is consistent with staff-approved ER-157-P and that the licensee adequately addressed the four additional requirements listed in the staff's safety evaluation (SE). The NRC staff also reviewed the power measurement uncertainty calculations to ensure: (1) the conservatively proposed uncertainty value of 0.37 percent correctly accounts for all uncertainties associated with power level instrumentation errors, and (2) the uncertainty calculations meet the relevant requirements of Appendix K to 10 CFR Part 50, as described in this section, above.

The licensee provided the following information about the Cameron LEFM CheckPlus System feedwater flow measurement technique and its implementation at Byron and Braidwood, Units 1 and 2. The Cameron LEFM CheckPlus System consists of an electronic cabinet (with its own cooling system) installed in the main steam line~~-~~ tunnel and measurement spool pieces installed in each of the four main feedwater flow lines upstream of the existing feedwater venturi flow meters. Each measurement section consists of 16 ultrasonic, multipath, transit time transducers, and feedwater pressure input.

The electronic cabinet controls magnitude, sequences transducer operations, makes time measurements, and calculates volumetric flow, temperature, and mass flow. The system software measures velocities at precise locations. The feedwater mass flow rate and temperature are transmitted to the plant process computer for use in calorimetric measurement of reactor thermal output. In the event of system failure the control room operators are alerted.

The UFM values for feedwater mass flow and temperature will be directly substituted for the existing venturi-based flow and ~~resistence~~resistance temperature detector (RTD) temperature inputs used in the plant calorimetric measurement calculations. The existing venturi-based feedwater flow and RTD temperature will continue to be used for other plant functions and may be used for plant calorimetric calculations in the event of a UFM failure.

Evaluation of Information provided in response to RIS 2002-03, Attachment 1, Section I, guidance Items A through H,

Items A through C

Items A, B, and C, in Section I of Attachment 1 to RIS 2002-03 guide licensees to identify the approved TRs, provide references to the NRC's approval of the measurement technique, discuss the plant-specific implementation of the guidelines in the TR, and identify the NRC staff's approval of the feedwater flow measurement technique.

In its June 23, 2011 letter (the submittal), the licensee identified Cameron TR ER-80-P, Revision 0, issued March 1997 , and its supplement, TR ER-157-P, Revision 8, and Revision 8, Errata, issued May 2008, as applicable to the Cameron LEFM CheckPlus System. The licensee also referenced the NRC approval of the TRs ER-80-P and ER-157-P in SEs dated March 8, 1999, and August 16, 2010.

The licensee stated in its submittal that the Cameron LEFM CheckPlus system will be permanently installed in Byron and Braidwood, Units 1 and 2, according to the conditions, limitations and qualifications of the August 16, 2010 SE approving ER-157P. The Cameron LEFM CheckPlus system was installed in Braidwood, Unit 1, during the spring of 2012. The Cameron LEFM CheckPlus system for Braidwood, Unit 2, was installed during the spring of 2011. The Cameron LEFM CheckPlus system for Byron, Unit No. 1, was installed during the spring of 2011. The Cameron LEFM CheckPlus system for Byron, Unit No. 2 was installed during the fall of 2011.

On the basis of its review of the licensee's submittals, as reflected in the above discussion, the NRC staff finds that the licensee has sufficiently addressed the plant-specific implementation of the Cameron LEFM CheckPlus system using proper TR guidelines. Therefore, the staff concludes that the licensee's description of the feedwater flow measurement technique and implementation of the power uprate using this technique contains the information requested by Items A through C of Section I of Attachment 1 to RIS 2002-03. This information was evaluated against the regulatory criteria in RIS 2002-03 and was found to be acceptable.

Item D

Item D in Section I of Attachment 1 to RIS 2002-03 guides licensees to address four criteria when implementing the feedwater flow measurement uncertainty technique. The NRC staff SEs on ER-80-P, and ER-157-P, both include these four plant-specific criteria, to be addressed by a licensee referencing these TRs for a PU. The licensee's submittal addresses each of the four criteria as described below.

1. Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.

   Licensee Response:

   > Implementation of the MUR power uprate license amendment will include developing the necessary procedures and documents required for continued calibration and maintenance of the LEFM system. Plant maintenance and calibration procedures will be revised to incorporate Cameron's maintenance and calibration requirements prior to raising power above the current licensed thermal power of 3586.6 MWt. The Braidwood

and Byron Technical Requirement Manuals (TRMs) will be revised to address contingencies for inoperable LEFM instrumentation.

A modification package has been developed for each installation outlining the steps to install and test the LEFM CheckPlus system. When each unit is shutdown for their respective refueling outages the LEFM CheckPlus systems will be installed. Following installation, testing will include an in-service leak test, comparisons of feedwater flow and thermal power calculated by various methods, and final commissioning testing. The LEFM CheckPlus system installation and commissioning will be performed according to Cameron procedures. Commissioning and start-up of the LEFM CheckPlus System will be performed by qualified Cameron personnel with site personnel assistance. The commissioning process provides final positive confirmation that actual field performance meets the uncertainty bounds established for the instrumentation. Final site-specific uncertainty analyses acceptance will occur after completion of the commissioning process.

The Braidwood and Byron, Units 1 and 2, LEFM CheckPlus system was calibrated in a site-specific model test at ARL. The testing at ARL provides traceability to National Standards. The spool piece calibration factor uncertainty is based on Cameron engineering reports. The calibration tests included a site-specific model of each of the Units hydraulic geometry.

Preventive maintenance will be performed based on vendor recommendations. The preventive maintenance program and LEFM CheckPlus system continuous self-monitoring feature ensure that the LEFM remains bounded by the ER-80P, as supplemented by ER-157P, analysis and assumptions. Establishing and continued adherence to these requirements assures that the LEFM CheckPlus system is properly maintained and calibrated. The preventive maintenance activities will be identified via the associated plant modification package. Typical activities performed include power supply checks, pressure transmitter checks, and clock verifications. Maintenance of the LEFM system will be performed by personnel who are qualified.

Instrumentation, other than the LEFM system, that contributes to the power calorimetric computation will be periodically calibrated and maintained using existing site procedures. Maintenance and test equipment, tolerance settings, calibration frequencies, and instrumentation accuracy were evaluated and accounted for in the thermal power uncertainty calculation.

Staff Evaluation and Conclusion

Because the description of the calibration and maintenance procedures (and associated documentation) states that Exelon will incorporate the appropriate Cameron requirements, the NRC staff concludes that the licensee has adequately addressed Criterion 1 (of plant-specific criteria identified in the SE on the TR).

2. For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed instrumentation and confirmation that the installed

instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in ER-80-P.

Licensee Response:

At the time of the submittal, only Byron, Unit No. 1, and Braidwood, Unit 2, had installed the LEFM CheckPlus systems. Based on the results of the modification and commissioning testing the LEFM CheckPlus systems as installed are in conformance with the analysis and assumptions given in Cameron's TR ER-80P, ER-157P, and the Byron and Braidwood unit specific "Bounding Uncertainty Analysis for Thermal Power Determination Reports", as well as the performance parameters identified in the Alden Laboratory Meter Factor Calculation and Accuracy Assessments. As of June 17, 2011, there has been no performance, operational, or maintenance issues that would indicate any non-conformance with the above.

NRC Staff Evaluation and Conclusion

The NRC staff became aware of a trending issue after June 17, 2011, through NRC oversight programs. Based on the NRC staff's evaluation of the trending issue in section 3.1.1.3 in this SE and consideration of the licensee's response described above, the NRC staff concludes that the licensee's response is adequate to address Criterion 2 and therefore meets the regulatory guidance.

3. Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Licensee Response:

Cameron has performed Unit specific bounding uncertainty analysis for Byron and Braidwood Stations, Unit 1 and 2. Copies of these analyses are provided in the licensee's June 23, 2011~~2~~, submittal. The calculations in these analyses are consistent with Cameron's TR ER-80P, as supplemented by ER-157P, ISA-P67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation" and Exelon standard NES-EIC-20.04. This approach has been approved by the NRC. The core thermal power uncertainty calculation which takes into account the uncertainty associated with the feedwater flow venturis is performed in accordance with Exelon standard NES-EIC-20.04 and is consistent with ISA-RP67.04.02-2000.

The fundamental approach used is to statistically combine inputs to determine the overall uncertainty. Channel statistical allowances are calculated for the instrument channels. Dependent parameters are arithmetically combined to form statistically independent groups, which are then combined using the square root of the sum of the squares approach to determine the overall uncertainty.

NRC Staff Evaluation and Conclusion

The NRC staff confirmed that the methodology used to calculate uncertainty as described above is based on accepted plant setpoint methodology, which meets the regulatory guidance for an approved method of meeting the relevant requirements of 10 CFR Part 50, Appendix K, as described in Section 2 of this SE. Therefore, the NRC staff concludes that the licensee has adequately addressed Criterion 3 and therefore meets the regulatory guidance.

4. Licensees for plant installations where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), should provide additional justification for use. This justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

   Licensee Response:

   Criterion 4 does not apply to Byron or Braidwood Stations, Units 1 or 2. Byron and Braidwood Stations LEFM CheckPlus systems were calibrated at Alden Research Laboratory. Cameron engineering reports for each of the Units evaluating the calibration test data from Alden Research Laboratory have been completed and have been provided. The calibration factors used for each Units LEFMs are based on the analysis contained in reports docketed as part of the LAR.

   NRC Staff Evaluation and Conclusion

   Because the LEFM installations at Braidwood and Byron, Units 1 and 2, were calibrated to unit specific piping configurations, the ~~Nyron~~ NRC staff agrees that Criterion 4 does not apply to Braidwood or Byron, Unit Nos. 1 or 2.

Item E

Item E in Section I of Attachment 1 to RIS 2002-03 guides licensees in the submittal of a plant-specific total power measurement uncertainty calculation, explicitly identifying all parameters and their individual contributions to the power uncertainty.

To address Item E of RIS 2002-03, in Attachments 8a-8d of the submittal, the licensee provided Cameron engineering reports ER-800, ER-801, ER-802, and ER-803. These reports provide calculations that demonstrate that the total thermal power uncertainty for the bounding unit is 0.37 percent. The licensee proposes that all units be uprated to the value of the most bounding plant.

Comment [LEF4]: Same comment as we previously provided in email dated 1/22/14. "As indicated in our June 23, 2011 submittal [Attachment 1 - §3.1 page 5 and §3.3 page 9, and Attachment 5 - §I.1.E page I-7) and verifiable in the four uncertainty reports (ER-800, ER-801, ER-802, and ER-803), the bounding uncertainty for all four Units is 0.345%. Exelon added additional margin to the 0.345% uncertainty to limit the requested power increase to be 1.63%."

In October 2011, the NRC staff reviewed and audited the calculations and determined that the licensee identified the parameters associated with the thermal power measurement uncertainty, provided individual measurement uncertainties, and calculated the overall thermal power

uncertainty. The licensee's calculations arithmetically summed uncertainties for parameters that are not statistically independent and statistically combined with other parameters. The licensee combined random uncertainties using the square root sum of squares approach and added systematic biases to the result to determine the overall uncertainty. This methodology is consistent with the vendor determination of the Cameron LEFM CheckPlus system uncertainty, as described in the referenced TRs, and is consistent with the guidelines in RG 1.105.

The NRC staff finds that the licensee has provided calculations of the total power measurement uncertainty at the plants, explicitly identifying parameters and their individual contribution to the power uncertainty. Therefore, the NRC staff concludes that the licensee has provided the information requested in Item E of Section I of Attachment 1 to RIS 2002-03, and this information is consistent with the RG 1.105, above.

Item F

Item F in Section I of Attachment 1 to RIS 2002-03 guides licensees to provide information to address the specified aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric.

In the submittal, the licensee addressed each of the five aspects of the calibration and maintenance procedures listed in Item F of RIS 2002-03:

1. Maintaining Calibration

The licensee stated that calibration will use procedures based on the appropriate LEFM CheckPlus technical manuals. Other calorimetric process instrumentation and computer points are maintained and calibrated using approved procedures.

2. Controlling Hardware and Software Configuration

The Cameron LEFM CheckPlus system is designed and manufactured in accordance with the vendor's quality assurance program. The licensee committed to maintaining, after installation, the software and hardware configuration by using existing procedures and processes, which include verification and validation of software configuration changes.

3. Performing Corrective Actions

Station personnel will monitor plant instrumentation that affects the power calorimetric input, including UFM inputs. Any problems detected will be handled according to the Station's corrective action process.

4. Reporting Deficiencies to the Manufacturer

The licensee states that any conditions found to be adverse to the quality of the LEFM CheckPlus system will be documented and reported to the vendor, as needed, to support corrective action.

Comment [LEH5]: This is different than what we provided. In the LAR (6/23/11), Attachment 5, Item 4 on page I-9 we provided "Deficiencies associated with the vendor's processes or equipment will be reported to the vendor as needed to support corrective action."

5. Receiving and Addressing Manufacturer Deficiency Reports

Cameron has procedures to notify users of important LEFM deficiencies. The licensee stated that it has existing processes to address the receipt of Cameron's deficiency reports.

On the basis of its review of the above statements, the NRC staff finds that the licensee has addressed the calibration and maintenance aspects of the Cameron LEFM CheckPlus System and all other instruments affecting the power calorimetric. Therefore, the staff concludes that the licensee has provided the information identified in Item F of Section I of Attachment 1 to RIS 2002-03, and that the information meets the regulatory guidance.

Items G and H

Items G and H in Section I of Attachment 1 to RIS 2002-03, guide licensees to provide a proposed allowed outage time (AOT) for the instrument and to propose actions to reduce power if the AOT is exceeded.

In the submittal, the licensee proposed a 72-hour AOT for operating above 3586.6 MWt (i.e., the current licensed thermal power limit) if the UFM becomes non-operational. The licensee specifically noted that any failure of the UFM, including failures of a single path or plane, will be treated as a complete failure of the UFM system and thus start the plant's allowable 72 hours to remain above 3586.6 MWt. In addition, the ability of the plant to stay above 358~~5~~56.6 MWt is contingent upon its ability to maintain steady-state conditions. If the plant should reduce power below 358~~5~~56.6 MWt during the AOT, the plant will not be permitted to return above that value until the UFM function is fully restored.

During the 72-hour AOT, in the event of a UFM failure, the plant would use the existing feedwater venturis for the calorimetric calculation. Because the FW venturis are regularly normalized to the UFM measurements, their measurements should be equivalent to the UFM over the 72-hour AOT. Use of calibrated feedwater venturis to remain at the uprated power level during the 72 hours that follow a UFM being declared non-operational is consistent with previous measurement uncertainty recapture submittals that the staff has approved. Venturi nozzle fouling and transmitter drift were considered as potential sources of error within the AOT window.

The basis for the 72-hour AOT includes the fact that there has been no evidence of feedwater venturi fouling at Braidwood and Byron, Unit Nos. 1 and 2. This conclusion was reached by Exelon based on a review of historical records and analysis of feedwater flow. Since there has been no observed feedwater venturi fouling, it is unlikely that fouling or sudden de-fouling will occur in the 72-hour AOT. In addition, Exelon analyzed the expected drift of the feedwater venturi and determined that this drift would be insignificant during a 72-AOT.

By letter dated November 1, 2011 Exelon provided information which allowed the NRC staff to understand the conformance of the implementation with respect to the TR, confirmed that the assumptions listed in the TR were valid for Braidwood and Byron MUR application, or described how plant-specific information was used in the application, confirmed that either a sufficiently large number of samples were used such that the two sigma values were 95/95 values, or Students-t was used as a multiplier for establishing the 95/95 confidence interval, and explained that number discrepancies resulted from misquoting the TR, but these errors did not affect the application since the numbers were provided for comparison purposes. The NRC staff determined that the responses were appropriate and resolved the staff's questions.

On the basis of its review of the licensee's submittals, the NRC staff finds that the licensee has provided sufficient justification for the proposed AOT and the proposed actions to reduce power level if the AOT is exceeded. Therefore, the staff concludes that the licensee has provided the

information requested by Items G and H of Section I of Attachment 1 to RIS 2002-03, and the information provided meets the regulatory guidance.

Conclusion

The NRC staff reviewed the licensee's proposed plant-specific implementation of the feedwater flow measurement device and the power uncertainty calculations and determined that the licensee's proposed license amendment is consistent with the staff-approved TR ER-157P. The NRC staff has also determined that the licensee adequately accounted for instrumentation uncertainties in the reactor thermal power measurement uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K, as described in Section 2 of this SE. The NRC staff finds the instrumentation and control aspects supporting the proposed thermal power to be acceptable.

3.1.1.2 Plant Specific Implementation of the Applicable Topical Reports
(RIS 2002-03, Attachment 1, Section I.1)

Regulatory Evaluation

Paragraph I.A of Appendix K to 10 CFR Part 50 allows licensees to justify a smaller margin for power measurement uncertainty. Licensees may apply the reduced margin to operate the plant at a power level higher than the previously licensed power. In the LAR, the licensee proposed to use a Cameron LEFM CheckPlus system to decrease the uncertainty in the measurement of feedwater flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.37 percent.

As discussed previously, the NRC determined that ER-80-P and ER-157-P, which describe the Cameron LEFM CheckPlus System for the measurement of feedwater flow, are an acceptable way of conforming to the regulations

General acceptance criteria apply to all aspects of testing in a certified facility, transfer from the test facility, initial operation, and long-term in-plant operation. These criteria are:

- Traceability to a recognized national standard. This requires no breaks in the chain of comparisons, all chain links must be addressed, and there can be no unverified assumptions,

- Calibration, and

- Acceptable addressing of uncertainty beginning with an initial estimate of the bounding uncertainty and continuing through all aspects of initial calibration in a certified test facility, transfer to the plant, initial operation, and long-term operation.

For CheckPlus, meeting these criteria includes documenting:

- Design and characteristics information,
- Calibration testing at a certified test facility,
- Any potential changes associated with differences between testing and plant operation including certification that initial operation in the plant is consistent with pre-plant characteristics predictions, and
- In-plant operation.

Technical Evaluation

Initial Design and Characteristics

To determine volumetric flow rate, the Caldon UFM transmits an acoustic pulse along a selected path and records the arrival of the pulse at the receiver. Another pulse is transmitted in the opposite direction and the time for that pulse is recorded. Since the speed of an acoustic pulse will increase in the direction of flow and will decrease when transmitted against the flow, the difference in the upstream and downstream transit times for the acoustic pulse provides information on flow velocity. Once the difference in travel times is determined, the average velocity of the fluid along the acoustic path can be determined. Therefore, the difference in transit time is proportional to the average velocity of the fluid along the acoustic path.

The CheckPlus provides an array of 16 ultrasonic transducers installed in a spool piece to determine average velocity in eight paths. The transducers are arranged in fixtures such that they form parallel and precisely defined acoustic paths. The chordal placement is intended to provide an accurate numerical integration of the axial flow velocity along the chordal paths. Using Gaussian quadrature integration, the velocities measured along the acoustic paths are combined to determine the average volumetric flow rate through the flow meter cross section. Note that this process assumes a continuous velocity profile in the flow area perpendicular to the spool piece axis. Although the velocity profile can be distorted, the distortion cannot be such that the Gaussian quadrature process no longer provides an acceptable mathematical fit to the profile, such as may occur if the profile is distorted in a way that is not recognized by the CheckPlus due to an upstream flow straightener.

To obtain the actual average flow velocity a calibration factor is applied to the integrated average flow velocity indicated by the UFM. The calibration factor for the Caldon UFMs is determined through meter testing at ARL and is equal to the true area averaged flow velocity divided by the flow velocity determined from the average meter paths to correlate the meter readings to the average velocity and hence to the average meter volumetric flow. The mass flow rate is found by multiplying the spool flow area by the average flow velocity and density. The mean fluid density is obtained using the measured pressure and the derived mean fluid temperature as an input to a table of thermodynamic properties of water. Typically, the difference between an uncalibrated CheckPlus and ARL test results is less than 0.5 percent. This close agreement means that obtaining a correction factor for a CheckPlus is relatively insensitive to error for operation under test conditions. Further, as discussed in this SE, correction factor is not a strong function of the difference between test and plant conditions and the same conclusion applies.

Use of a spool piece and chordal paths improves the dimensional uncertainties including the time measurement of the ultrasonic signal and enables the placement of the chordal paths at precise locations generally not possible with an externally mounted UFM. This allows a chordal UFM to integrate along off-diameter paths to more efficiently sample the flow cross section. In addition, a spool piece has the benefit that it can be directly calibrated in a flow facility, improving measurement uncertainty compared to externally mounted UFMs that were historically installed in nuclear power plant feedwater lines.

The NRC staff concludes that the applicant acceptably meets the guidance as approved in the NRC staff's SE dated August 16, 2010, for the above aspects of its proposed use of CheckPlus UFMs. Flow straighteners are not used immediately upstream of the installations and other

potential distortions of the flow profile are either absent or acceptably addressed in ARL testing. Coverage of other aspects of the proposed use is addressed in other sections of this SE.

Test Facility Considerations

Test Facility Qualification

Calibration testing at a qualified test facility and at the plant involves traceability to a national standard, facility uncertainty, and facility operation. At www.aldenlab.com, ARL states that "Alden is the largest independent supplier of National Institute of Standards and Technology (NIST) traceable Flow Meter Calibration Services in the country." The NRC staff audited testing at ARL in February 2006 and verified ARL's statement with respect to traceability to NIST (ADAMS Accession No. ML11311A174). The NRC's audit found that ARL's processes and operation were consistent with the claimed facility uncertainties. The NRC staff also observed testing during a visit to ARL in August 2009, and observed some improvements in test facility hardware. The NRC staff concluded that these changes would not change its previous conclusions during the February 2006 audit regarding test operations and results. In Attachments 8a-8d of the licensee's June 23, 2011, submittal, Cameron restated that "all elements of the lab measurements...are traceable to NIST standards." Based on the above, the NRC staff concludes that that ARL meets the stated testing criteria.

All CheckPlus installations to date have been calibrated at ARL. The audit also confirmed that ARL was providing acceptable test data for the configurations under test. Consequently, the qualification of ARL does not need to be investigated further or confirmed with respect to CheckPlus testing provided test conditions remain consistent with the referenced conditions.

Test Fidelity and Test Range

Test fidelity, such as test versus planned plant configuration, test variations to address configuration differences, and potential effects of operation on flow profile and calibration, should be addressed on a plant-specific basis. Applicant requests must provide a comparison of the test and plant piping configurations with an evaluation of the effect of any differences that could affect the UFM calibration. Further, sufficient variations in test configurations must be tested to establish that test-to-plant differences have been bracketed in the determination of UFM calibration and uncertainty. Historically, calibration testing has acceptably covered upstream effects by applying a variation of configurations to distort the flow profile. Further, if the spool piece may be rotated during plant installation from the nominal test rotation, the effect of rotation should be addressed during testing.

Plant piping configuration drawings must, at a minimum, include isometrics with dimensional information that describe piping, valves, feedwater flow meters, and any other components from the feedwater pumps to at least 10 pipe diameters downstream of the feedwater flow meter that is most distant from the feedwater pump. Preferable are scale three dimensional (3D) drawings in place of isometrics that show this information. Test information must include 3D drawings of the test configuration including dimensions.

Reference O provides test configuration. The licensee provided an attachment of the typical installation locations drawings for the LEFMs at each of the four units. The LEFMs are installed upstream of the venturis in the plants and downstream of the common feedwater header. There are no flow straighteners being installed. Distances between the exit of the CheckPlus spool pieces and the downstream venturis in the tests and plant are sufficient such that there will be

no effect on the LEFM. As discussed in the section of ~~e~~Evaluations of the ~~e~~Effect of Downstream Piping Configurations on Calibration, below, the NRC staff concludes that this separation distance is large enough such that there will be no effect on UFM calibration and the facility uncertainty is acceptable.

Weigh tank tests were run at different flow rates for each simulated feedwater loop. Tests included 100 percent and lower flow rates through the CheckPlus and some tests included an eccentric orifice upstream in the feedwater pipes containing the CheckPlus. Most test results were included in the reported main FW calculation.

Transfer from Test to Plant and In-Plant Installation

Each applicant for a PU must conduct an in-depth evaluation of the UFM following installation at its plant that includes consideration of any differences between the test and in-plant results and must prepare a report that describes the results of the evaluation. This should address such items as calibration traceability, potential loss of calibration, cross-checks with other plant parameters during operation to ensure consistency between thermal power calculation based upon the LEFM and other plant parameters, and final commissioning testing. The process should be described in written documentation and a final commissioning test report should be available for NRC inspection.

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The NRC notes that, to date, the only UFM calibration traceability associated with transfer from the test facility to United States nuclear power plants that has been acceptably demonstrated is that provided by the Check and CheckPlus UFMs due to the ability to provide the flow distribution/velocity profile as a function of radius and angular position in the spool piece, the small calibration correction necessary to fit test data to UFM indication, and the demonstrated insensitivity to changes in operation associated with transfer changes and plant changes. Although other means have been used to obtain flow rate, such as use of tracers in the feedwater, they have not been demonstrated to provide the small uncertainty obtainable with a CheckPlus. Experience to date is that a UFM must provide flow profile information and calibration traceability when extrapolating from test flow rate and temperature conditions to plant conditions. Transfer uncertainty is associated with any changes due to installation in the plant such as mechanical and operating conditions. Mechanical perturbations include such items as transducer installation, mechanical misalignment, and fidelity between the test and plant. Changes in operating conditions involve consideration of such potential effects as noise due to pumps and valves, changes in flow profile, including swirl, flow rate, and temperature.

As previously identified, the test facility configuration and test parameters are expected to provide a basis for providing fidelity between the test and plant. However, an exact correspondence is probably not possible. Potential differences are expected to have been addressed and the UFM is expected to provide a capability to both identify differences and to address them during operation.

The applicant addressed uncertainty in Appendices 8a-8d of the submittal. As stated in the section on Test Fidelity and Test Range**,** above, the facility uncertainty is acceptable. Appendices 8a-8d of the submittal ~~is~~ are referenced for transducer installation uncertainty. The content is essentially identical to Appendix D of Cameron's Engineering Report ER-157(NP-A), Revision 8 that the NRC staff found acceptable in its October 15, 2010, SE (ADAMS Accession No. ML102950252 for both documents). Based on the above, the NRC staff concluded that the licensee's treatment of transducer installation uncertainty is acceptable. The licensee showed that LEFM commissioning will include verification of ultrasonic signal quality and evaluation of

actual plant hydraulic flow profiles as compared to those documented during the ARL testing. These parameters will be incorporated as required into the LEFM during commissioning. This meets the regulatory guidance as approved in the NRC staff's October 15, 2010, SE. The NRC staff therefore concluded that it is acceptable.

In-Plant Operation

Many of the calibration aspects associated with transfer from a test facility to the plant apply during operation as valve positions change, different pumps are operated, and physical changes occur in the plant. The latter include such items as temperature changes, preheater alignment and characteristics changes, pipe erosion, pump wear, crud buildup and loss, and valve wear. Further, potential UFM changes, such as transducer degradation or failure, may also occur and the UFM should be capable of responding to such behavior. Either the UFM must remain within calibration and traceability must continue to exist during such changes or the UFM must clearly identify that calibration and traceability are no longer within acceptable parameters. Experience is that the CheckPlus is capable of handling these operational aspects. Further, as stated above, UFM operation should be cross-checked with other plant parameters that are related to feedwater (FW) flow rate. Should such checking identify abnormal behavior, it should be identified to the NRC, the validity of the final commissioning test report should be confirmed, and the final commissioning test report should be updated as necessary to reflect the new information. Further, the UFM must be considered inoperable if its calibration is no longer established to be within acceptable limits.

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Comment [LEH6]: Does this cross-checking refer to the cross checking mentioned in the previous section? (See highlighted wording) If so, does the action stipulated pertain to the action in that section or is the intent that these are ongoing action requirements?

Clarification is requested as to the NRC's intent for the stipulated action requirements.

These actions were not presented by us in our license submittal documentation.

Section I.1 of Reference V provides coverage of training, calibration, maintenance, procedures, entry into the corrective action program, and procedures to ensure compliance with the requirements of 10 CFR Part 50, Appendix B. Based on the above, the NRC staff concluded that in-plant operation meets regulatory guidance and is acceptable.

Operation with a ~~f~~Failed Component

A brief description should be provided that covers system self-testing features, channel checks, control room alarms, and plant process computer functions. The following should be addressed to cover conditions if the CheckPlus system becomes degraded or inoperable:

(1) Operator response.

(2) Changes in FW flow input to the core thermal power calculation.

(3) Allowed outage time (AOT)[2]: Time when continued operation at full power is permitted and time when power must be reduced, including specification of the reduced power level.

(4) Justification for the AOT with respect to such topics as calibration of FW venturis, venturi fouling or defouling, monitoring of other indications of core thermal power such as average power range monitors, steam flow rate, feed flow rate, turbine first stage pressure, and main generator output.

(5) Response if the plant computer system is not operable.

[2] The NRC has typically approved an AOT of 72 hours if acceptably justified.

Cameron report ER-157-P, Revision 8, states:

> [t]he redundancy inherent in the two measurement planes of an LEFM CheckPlus also makes this system more resistant to component failures.

when compared to the Check.

> For any single component failure, continued operation at a power greater than that prior to the uprate can be justified with a CheckPlus system…since the system with the failure is no less than an LEFM Check.

This is acceptable subject to two qualifications:

(1) Continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time are plant specific and must be acceptably justified.

(2) The only mechanical difference that potentially affects the quoted statement is that the CheckPlus has 16 transducer housings interfacing with the flowing water whereas the LEFM Check has 8. Consequently, a CheckPlus operating with a single-failure is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if an applicant wishes to operate as stated. An acceptable quantification method is to establish the effect in an acceptable test configuration such as can be accomplished at ARL.

Sections I.1.G-I.1.H of the submittal address~~es~~ allowed outage time, monitoring of CheckPlus status, and operational processes associated with a degraded or non-operational CheckPlus. The difference between a degraded CheckPlus and a Check is covered by the ARL test results.

To operate above the current license thermal power (CLTP) of 3586.6 MWt, the licensee proposes to use the Cameron LEFM CheckPlus System in the normal mode. In the normal mode of operation, both planes of transducers are in service and system operations are processed by both central processing units (CPUs). The LEFM reading will be used as input for the FW flow in the calorimetric. The licensee plans to respond to single-path or single-plane failures in the same way that they will respond t~~w~~o a whole system failure. In the event of a failure of a single-path or single-plane or other system failure, the licensee will declare the system inoperable. The requirement will be to restore the system to operable within 72 hours or the plant will be required to reduce power to the CLTP or 3586.6MWt any time the plant is above the CLTP. If the plant is below the CLTP of 3586.6MWt when the LEFM is declared inoperable, the plant will not be allowed to increase power above the CLTP of 3586.6MWt.

The licensee provided justification for the proposed 72-hour Technical Requirements Manual (TRM) technical limiting condition operation (TLCO) action time. This justification included the statements as follows:

> There has been no evidence of feedwater flow venturi fouling at Byron or Braidwood Stations. This is based on a review of historical work orders that document this observation as part of a procedural requirement performed every refueling outage. In addition, historical data was gathered over the last several years where feedwater venturi flow was analyzed. This data indicated that there was no divergence in feedwater flow indication that would suggest venturi

> fouling…a typical power measurement uncertainty calculation for a two-loop PWR is shown to be approximately 1.4%. The systematic error associated with feed flow nozzle differential pressure in this calculation is shown to be approximately 1.0%. Assuming this was calculated based on an 18-month cycle; this would represent a maximum potential drift in the differential pressure measurement of less than 0.002% per day. Over a 72-hour period, this would have an insignificant effect on the feedwater flow measurement. Feedwater flow differential pressure instrument drift history at Byron and Braidwood is consistent with the assumptions of this typical calculation.

The NRC staff agrees with the above statement that under normal operating conditions the drift of a venturi over a 72-hour period would be minimal. The reference above as well as previous precedent of plants with similar operating experience and approved outage time provides reasonable assurance that the plant will operate safely for the 72-hour outage time and maintain the licensed power level.

These actions are to be covered in the LEFM operability requirements contained in the Byron and Braidwood TRM TLCO. Based on the above, the NRC staff concludes that operation with an inoperable (non-functional) CheckPlus has been acceptably addressed and planned operation with a failed CheckPlus component has been acceptably addressed.

Spool Piece Dimensional Effects on UFM Response

Appendix A of ER~~P~~-157-P addresses the effect of variation in such spool piece dimensions as as-built internal diameter and sonic path lengths, path angles, and path spacing. The licensee's description for addressing these effects is acceptable.

Transducer Installation Sensitivity

Transducers may be removed after ARL testing to avoid damage during shipping the spool piece to the plant. Further, transducers may be replaced following failure or deterioration during operation. Replacement potentially introduces a change in position within the transducer housing that could affect the chordal acoustic path. Appendix D of ER-157-P addresses replacement sensitivity by describing tests performed at the Caldon Ultrasonics flow loop and provides a comparison of test results to analyses of potential placement variations that shows that the test results are bounded by predicted behavior. The NRC staff concludes that there could be an uncertainty associated with the test loop even if nothing were changed. This is not addressed in the ER-157-P information. Rather, all of the test uncertainty is conservatively assumed to be due to transducer replacement. Further, as stated, the analyses predict a larger uncertainty than obtained during testing, and the analysis uncertainty is used for transducer replacement uncertainty. The NRC staff concludes that this approach is sufficient to cover the inability of the test loop to achieve flow rates comparable to those obtained in plant installations and to cover any analysis uncertainty associated with applications with pipe diameters that differ from the tests. Consequently, the NRC staff ~~f~~concludes that the transducer replacement has been acceptably addressed and the ER-157-P, Revision 8, process for determining transducer replacement uncertainty is acceptable.

The Effects of Random and Coherent Noise of LEFM CheckPlus Systems

Appendix C of ER-157-P provides a proprietary methodology for test- and plant-specific calculation of the contribution of noise to CheckPlus uncertainty. The NRC staff's SE dated August 16, 2010, concluded that applicants may use this methodology in their MUR requests.

The licensee's June 23, 2011, submittal demonstrates that critical performance parameters, including signal to noise ratio, are continually monitored for every individual meter path and alarm setpoints are established to ensure corresponding assumptions in the uncertainty analysis remain bounding. Signal noise will be minimized via strict adherence with Cameron design requirements. LEFM commissioning included verification of ultrasonic signal quality.

In Appendices A.3 to Appendices 8a-8d of the submittal, the applicant reported test signal to noise ratios for random and coherent noise that were within the assumptions in the Cameron Topical Report ER-80-P, and that uncertainty attributable to the electronics and signal to noise ratio are included in the overall meter factor uncertainty.

The NRC staff concluded that the test results and the above Appendices 8a-8d of the submittal coverage of noise are sufficient to ensure that this topic is acceptably addressed.

Evaluation of the Effect of Downstream Piping Configurations on Calibration

The turbulent flow regimes that exist when the plant is near full-power result in limited upstream flow profile perturbation from downstream piping. Consequently, the effects of downstream equipment need not be considered for normal CheckPlus operation provided changes in downstream piping, such as the entrance to an elbow, are located greater than two pipe diameters downstream of the chordal paths. However, if the CheckPlus is operated with one or more transducers out of service, the acceptable separation distance is likely a function of transducer to elbow orientation. In such cases, if separation distance is less than five pipe diameters, it should be addressed.[3] [4]

As discussed in the section on Test Fidelity and Test Range, above, separation from downstream components is needed so that CheckPlus operation will not be affected. The NRC staff concludes that the in plant separation from downstream piping components such as elbows and venturis is acceptable and will not affect CheckPlus operation.

Evaluation of Upstream Flow Straighteners on CheckPlus Calibration[5]

Operation with an upstream flow straightener is known to affect CheckPlus calibration to a greater extent than most other upstream hardware. If a licensee proposes this configuration, it must provide justification.

A previously undocumented effect of upstream tubular flow straighteners on CheckPlus calibration was discovered during ARL testing while NRC staff members were at the site on August 24, 2009, that did not appear to apply to any previous CheckPlus installations. As follow-up, additional tests were conducted with several flow straighteners and two different pipe /spool piece diameters to enhance the statistical data basis and to develop an understanding of

[3] This was the case, for example, with a Calvert Cliffs application that the NRC found acceptable (References Q and R). In that installation, the distance between the spool piece exit is 15 inches from the downstream elbow and the chordal paths are 2.7 diameters upstream of the entrance to the piping bend.
[4] Although this is not addressed in ER-157P Rev. 8, it is addressed in the August 16, 2010 NRC SE (ADAMS Accession No. ML102160663).
[5] This is not addressed in ER-157P Rev. 8 but is addressed in the August 16, 2010 NRC SE.

the interaction between flow straighteners and the CheckPlus. The results are provided in ER-790-P, Revision 1 (ADAMS Accession No. ML100840026). It should be noted that the results do not apply to the Check UFM. Consequently, the findings do not apply to a Check that is installed downstream of a tubular flow straightener.

Cameron concluded that two additional meter factor uncertainty elements are necessary if a CheckPlus is installed downstream of a tubular flow straightener and provided uncertainty values derived from the test results. The data also provide insights into the unique flow profile characteristics downstream of tubular flow straighteners and a qualitative understanding of why the flow profile perturbations may affect the CheckPlus calibration.

Cameron determined that the two uncertainty elements are uncorrelated and therefore combined them as the root sum squared to provide a quantitative uncertainty. The NRC staff concludes that the Cameron approach is valid, but is concerned that the characteristics of existing tubular flow straighteners in power plants may not be adequately represented by samples tested in the laboratory. Any applicant that requests an MUR with this configuration should provide justification for claimed CheckPlus uncertainty that extends the justification provided in ER-790-P.

The NRC staff concludes that, because there are no flow straighteners installed in the applicant's feedwater lines and flow straightener effects are not a concern.

Other Thermal Power Calculation Considerations

Steam Moisture Content